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OAK RIDGE  
NATIONAL  
LABORATORY

*Letter Report*

FMDP LWR PEIS Data Report  
Rev. 3

MARTIN MARIETTA

December 21, 1995

Fissile Materials Disposition Program  
Reactor-Based Technologies  
Facility Project

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.







## **APPENDIX A**

### **COMMERCIAL LIGHT-WATER REACTOR OPERATIONAL CHARACTERISTICS**



## **APPENDIX A**

### **COMMERCIAL LIGHT-WATER REACTOR OPERATIONAL CHARACTERISTICS**





## GLOSSARY

TABLE A – COMMERCIAL REACTOR OPERATIONAL CHARACTERISTICS

**Thermal Rating (MWt)** – The maximum thermal power of the reactor authorized by the NRC, expressed in megawatts. (Reference 1)

**Average Thermal Generation (MWh)** – The average annual thermal energy of the unit during the periods 1988 to 1992 as measured or computed by the licensee in megawatt hours. (Reference 2)

**Electric Rating (MWe gross)** – The gross electrical output of the unit specified by the utility and used for the purpose of plant design. (Reference 1)

**Average Electricity Generated (MWh)** – The average annual electrical output of the unit during the periods 1988 to 1992 as measured at the output terminals of the turbine generator, in megawatt hours. (Reference 2)

**Average Unit Availability Factor (%)** – The total clock hours in the year that the unit operated on-line or was capable of such operation (unit reserve shutdown hours + hours generator on-line) averaged over the period 1988 to 1992. (Reference 2)

**Cooling Type** – The type of cooling used by the unit to remove waste heat from the facility. (Reference 1)

## Cooling Type

- 
1. Once Through
  2. Recirculating
    - A. Cooling Towers
      1. Natural Draft
      2. Mechanical Draft
    - B. Cooling Lake(s), Pond(s),  
or Canal(s)
  3. Closed Cycle Cooling Reservoir
- 

**Water Uptake (m<sup>3</sup>/sec)** – Cooling water flow rate at the intake of the main condensers in cubic meters per second. (Reference 1)

**Site Size (Acres)** – The size of the area dedicated to the facility in acres. (One site is on a government reservation so the total area dedicated to the facility was calculated from the acreage within the 1.21 mile exclusion zone.) (Reference 1)

**Estimated Population in Year 2010 Within 50 Miles** – An estimate of the population within a 50-mile radius of the facility for the year 2010 based upon the 1980 census and predicted growth rates for the location. (Reference 1)

**Gaseous Effluents** – Amount of the selected radionuclides released to the atmosphere from all sources as calculated by the licensee reported in curies and averaged for each plant over the period 1988 to 1991. (Reference 3)

**Liquid Effluents** – Amount of the selected radionuclides released from the facility as calculated by the licensee reported in curies and averaged for each plant over the period 1988 to 1991. (Reference 3)

**Radioactive Waste** – The amount of low level radioactive solid waste shipped from the facility to a licensed disposal site per year and the number of such shipments per year averaged for each plant over the period 1988 to 1992. (Reference 3)

**Stored Mixed/1000 MWe ( $\text{m}^3/\text{year}$ )** – The amount of mixed radioactive waste in cubic meters per year that is stored on the plant site per 1000 MW of electrical output. This number is an average for all plants in the United States. (Reference 1)

**Annual Whole Body Dose for All Personnel (REM)** – The amount of radiation dose in REM that was received at each plant for all persons (station, utility, and contractor) categorized by job function and averaged over the period 1988 to 1992. (Reference 4)

**Annual Whole Body Dose for Refueling Personnel (REM)** – The amount of radiation dose in REM that was received at each plant for refueling personnel only categorized by job function and averaged over the period 1988 to 1992. (Reference 4)

**Refueling Outages (1988–1992)** – The number of refueling outages that occurred for the plant for the period 1988 to 1992. (Reference 2)

**Average # Assemblies Discharged** – The number of fuel assemblies that were discharged to the spent fuel pool or other on-site storage facility. This value was calculated using an average of one-third of the core for PWRs and one-fourth of the core for BWRs based on the total number of elements in the core reported in Reference 5.

**Licensed Spent Fuel Pool Storage Capacity (# Assemblies)** – The total number of fuel assemblies that may be stored in the spent fuel pool as authorized by the plant license. (Reference 5)

**Note 1 –Projected Date for Last Refueling Discharge to Spent Fuel Pool** – Dates for the projected last refueling discharge to spent fuel pool are not presented since data has not been collected by the NRC for this item since 1990 and many utilities have received approval to re-rack their spent fuel pools, utilize dry storage, or ship spent fuel either to a repository or to another reactor site.

## References

1. U.S. Nuclear Regulatory Commission, *Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants, Draft for Comment*, NUREG-1437, Vol. 2, August 1991.
2. U.S. Nuclear Regulatory Commission, Annual Reports (1998–1992), *Licensed Operating Reactors, Status Summary Report*, NUREG-0020, Vols. 12–16.
3. U.S. Nuclear Regulatory Commission, Annual Reports (1998–1991), *Radioactive Materials Released from Nuclear Power Plants*, NUREG/CR-2907, Vols. 9–12.
4. U.S. Nuclear Regulatory Commission, Annual Reports (1998–1992), *Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities*, NUREG-0713, Vols. 10–14.
5. Individual plant *Final Safety Analysis Reports*.

Table A - COMMERCIAL REACTOR OPERATIONAL CHARACTERISTICS

27-Feb-85		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		BWR		B	
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## **APPENDIX B**

### **COMMERCIAL LIGHT-WATER REACTOR USING MIXED OXIDE FUEL (OPERATIONAL DELTAS)**



## APPENDIX B – PHYSICAL AND OPERATIONAL CHANGES

*Provide descriptions of the physical and operational changes that would be made for a typical reactor using mixed oxide fuel in lieu of conventional low enriched uranium fuel.*

### NOTES:

- (1) The baseline values quoted below are based on Table A in Appendix A. The original Table A provided by Tetra-Tech has been extensively modified by ORNL. The major modification involved a reformulation of the table to include eight PWRs and four BWRs. All units are comparable in size and operating characteristics. Where possible, operational data was obtained for five years and averaged for each reactor. These averages correspond to the new "generic" values.
- (2) The numbering system employed for the tables in Appendix B table are matched to the section sections which contain them.

## PHYSICAL CHANGES TO SITE AND REACTOR SYSTEMS

### External Appearance and Site

A reactor using MOX fuel may have an addition to the fuel receiving and storage buildings that could alter the "profile" of the plant to a small degree.

No significant additional land would be needed for an existing plant using MOX fuel.

### Reactor Systems

Based on a review of the referenced vendor reports, use of MOX fuel in a PWR or BWR would not require modification or enlargement of the containment structure. (This assumes that the accident performance of the MOX-fueled plant is not significantly different from that of the UO<sub>2</sub>-fueled plant).

### Cooling and Auxiliary Water Systems

The design and operation of cooling and auxiliary water systems would not be affected by use of MOX fuel in either a PWR or BWR.

### Radioactive Effluent Treatment Systems

A summation of the solid, gaseous and liquid radioactive effluents expected from operation with MOX fuel is contained in Sections B.3, B.5, and B.6 respectively.

### Transportation of Radioactive Materials

Section B.9 contains a description of the transportation impacts for transport of MOX fuel.



## B.1 OPERATIONAL EMPLOYMENT CHANGES

*Provide estimates of the operational employment changes for a typical commercial nuclear power facility.*

### Vendor Employment Estimates

Estimates of the operational employment changes for a typical commercial nuclear power facility are based on the following vendor-supplied information from the DOE PDS:

Based on a review of the three referenced vendor studies, operational employment increases for use of MOX fuel are expected to be minimal. ABB-CE, identified a possible need for additional security personnel for receiving and storing MOX fuel on-site: 15 for a single unit plant; 28 for a dual unit. GE and Westinghouse indicated a minimal number of additional personnel would be needed for operations.

Table B.1.1. Vendor Employment Data

Vendor	Report Pages	Additional operating personnel for MOX fuel
ABB-CE	Ref. 1 6-12, 13	Single unit: 15 security staff Two Units: 28 security staff
GE	Ref. 2 6.3-5, 6	Case: <u>3 REACTORS</u> Table 6.3-2: BWR Plant Incremental O&M Costs for Full MOX Core Operations (Over and above normal O&M cost levels for LEU operation - 1/93 base date)  Onsite Staff 3020 K\$/year Offsite Tech Support <u>868</u> 3888 K\$/year (about 40 FTE)  Table 6.3-3: Estimate of O&M Costs for MOX and Spent Fuel Storage Facility-3 unit reference case  Spent Fuel 93.6 K\$ MOX Fuel 64.4 K\$ Security <u>177.4</u> K\$ 335.4 K\$ single unit (~4 FTE) 3 unit (~9 FTE)  - Additional security during refueling met through overtime
<u>W</u>	Ref. 3 6.3-1	MCA* personnel: 2 @ \$130,000/year

\*Material Control and Accountability

Based on the figures above, estimates of the maximum operational employment changes for a typical existing nuclear power plant using MOX fuel are expected to be as follows:

**Table B.1.2. Operational Employment Changes**

<b>Number of Reactors at Site</b>	<b>Typical Plant Employment</b>	<b>Plant Using MOX</b>	<b>Delta</b>
1 unit	832	847	+15
2 unit	1247	1275	+28
3 unit	2404	2439	+35*

\*For a 3-unit plant, it is assumed that security needs would be met by about half the delta required for a two-unit site over a single unit, i.e., 13 additional staff for a two-unit over a single unit; seven additional for a three-unit site over a two-unit site. During fuel receiving, additional security requirements would be supplemented by overtime.

## References

1. "Analysis of Existing ABB-CE LWRs for the Disposition of Weapons-Grade Plutonium," DOE Plutonium Disposition Study, Combustion Engineering, Inc., Windsor, Connecticut, June 1, 1994.
2. "Study of Plutonium Disposition Using Existing GE BWRs," NEDO-32361, GE Nuclear Energy, June 1, 1994.
3. "Plutonium Disposition in Existing PWRs," DOE/SF/19683-6, Westinghouse Electric Corporation, June 1, 1994.

## B.2 CORE SOURCE TERM CHANGES

*Provide a description of the physical changes necessary in the reactor core for a reactor burning mixed oxide fuel.*

### B.2.1. Background

This Section provides information concerning the end of cycle radiological inventory of a typical  $\text{UO}_2$  fueled core and summarizes expected changes in those inventories due to the introduction of MOX fuel. It is our understanding that the MACCS code will ultimately be used to analyze accident source terms and their associated consequences. Since some accident source terms in Section B.8 are described in terms of release fractions, the initial core inventory must be supplied to provide the input necessary to run the MACCS code (Ref. 1). This Section will provide an inventory for 60 nuclides that are to be used in tandem with the release fractions. In addition, this Section provides estimates of MOX/ $\text{UO}_2$  total core inventory ratios. These ratios can be utilized in sections B.5, B.6, and B.8 to quantitatively assess the differences in emissions between a MOX fueled plant and a  $\text{UO}_2$  fueled plant.

### B.2.2. Core Inventories for a $\text{UO}_2$ Fueled Reactor

There are a number of literature sources that provide estimates of the core inventories for a PWR or BWR. Specifically, the objective here is to provide the end of equilibrium cycle inventories for 60 nuclides that are important for modeling the consequences resulting from both non-severe and severe accidents. The core inventory described in this Section is appropriate for a "generic existing light-water reactor" for use in the PEIS. The ORIGEN (either ORIGEN2 or ORIGEN-S) code is generally used to provide this information for both PWRs and BWRs. Critical input parameters that significantly affect the calculation of core inventory are burnup (MW-D/MT), core neutron spectrum, core thermal power level, and the loadings of both structural materials and uranium oxide fuel.

Table A (from Appendix A) reports an average thermal power of 3404 MWth for the 12 large reactors that were selected to serve as the experience base. Reference 2 provides a description of the plant and the inventories for a large PWR rated at 3411 MWth. The inventory for a baseline  $\text{UO}_2$  fueled reactor was chosen from Table 7-2 of Ref. 2 and was extracted from an ORIGEN2 calculation. The inventories taken from Ref. 2 were converted to curies and are listed in Table B.2.1. The difference in power levels between the 3411 MWth rating and the average value 3404 MWth in Table A is negligible. The average discharge burnup for a spent fuel assembly is assumed to be 51,700 MW-D/MT. The core is assumed to consist of three batches that were irradiated to 17,200, 34,500, and 51,700 MW-days per metric tonne.

The values in Table B.2.1 were compared to the corresponding information for an existing large BWR found in Table 2.3-1 of Ref. 3. This comparison is useful both from the standpoint of an "independent" check and to see if any significant differences might exist between large PWRs and BWRs. The comparison showed that there is excellent agreement between the two inventories. For the purposes of this study, it was concluded that Table B.2.1 represents a reasonable nominal inventory for a light-water reactor.

### B.2.3. MOX/ $\text{UO}_2$ Core Inventory Ratios

Differences in the nuclide inventories between a MOX core and a  $\text{UO}_2$  core have been examined using several sources of information. The most complete listing of a MOX vs  $\text{UO}_2$  comparison is found in Ref. 4. This listing provides 60 isotopic inventories (a core using MOX and a core using  $\text{UO}_2$ ) for an *advanced PWR* plant with a 1993 megawatt thermal rating. A ratio of the isotopic MOX core inventory to the inventory of a  $\text{UO}_2$  fueled core was obtained from Table 1.3.4-3 of Ref. 4. These ratios are shown in Table B.2.2; the 60 isotopes in the upper portion of the table corresponds to the list in Table B.2.1.

The tritium, Xe-131m, and Xe-133m listings in Table B.2.2 were added because these isotopic ratios are needed to support future radiological consequence calculations using the information in sections B.5, B.6, and B.8. For tritium, the ratio was chosen based on the work documented in Ref. 5. A ratio of 1.09 can be calculated from Table IV C-13 of this reference. It must be remembered, however, that the calculations in that reference utilized *recycled MOX from reactors*, not weapons grade MOX and were done for a large BWR.

The values for Xe-131m and Xe-133m in Table B.2.2 were simply assumed to be equivalent to ratios of Te-131m and I-133, respectively. Surrogates for Kr-83m, Xe-135m, and Xe-138 were chosen to be Kr-85, Xe-135, and Ba-139, respectively. These isotopes were chosen as comparisons because of the similarities of their atomic weights which result in the proximity of their fission yield fraction on the plutonium fission yield curve. At present, no explicit information on these ratios has been found in the literature.

The values in Table B.2.2 have been compared to the MOX/UO<sub>2</sub> ratios calculated from information (for an existing BWR) found in Table 2.3-1 of Ref. 3. This comparison shows that most of the ratios are very similar. The most notable differences are I-134 (0.96 vs 1.24) and Cs-134 (0.65 vs 1.08). While these differences may seem significant, it is important to assess the possible uncertainty associated with the numbers in Tables B.2.1 and B.2.2 in line with their usage. Most applications of these inventories and ratios (for estimates of accident releases and emissions) have uncertainties in other aspects of the analysis that far overshadow the uncertainties associated with either the inventory or MOX/UO<sub>2</sub> ratios presented in this section.

## References

1. Chanin, D. I. et al., *MELCOR Accident Consequence Code System (MACCS) User's Guide*, NUREG/CR-4691, SAND86-1562, Vol. 1 and 2, Sandia National Laboratories, Albuquerque, NM, February 1990.
2. Oak Ridge National Laboratory and Resources for the Future, draft of May, 1993. *Damages and Benefits of the Nuclear Fuel Cycle: Estimation Methods, Impacts, and Values*, to be published in March 1995.
3. GE Nuclear Energy, *Study of Plutonium Disposition Using Existing GE Advanced Boiling Water Reactors*, NEDO-32361, June 1, 1994.
4. Westinghouse Electric Corporation *PDR600 Plutonium Disposition Study*, DOE/SF/19683-1, May 15, 1993.
5. U.S. Nuclear Regulatory Commission, *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light-Water Cooled Reactors*, NUREG-0002, Vol. 3, August 1976.

Table B.2.1. Core Inventory for a Typical UO<sub>2</sub> Fueled Light-Water Reactor<sup>a</sup>

Isotope	Curies	Isotope	Curies
Am-241	9.76E+03	Pu-238	2.80E+05
Ba-139	1.64E+08	Pu-239	3.07E+04
Ba-140	1.57E+08	Pu-240	4.01E+04
Ce-141	1.52E+08	Pu-241	9.41E+06
Ce-143	1.40E+08	Rb-86	1.95E+05
Ce-144	1.07E+08	Rh-105	7.99E+07
Cm-242	2.84E+06	Ru-103	1.33E+08
Cm-244	2.36E+05	Ru-105	8.62E+07
Co-58 <sup>b</sup>	4.96E+05	Ru-106	3.91E+07
Co-60 <sup>b</sup>	6.53E+05	Sb-127	9.17E+06
Cs-134	1.51E+07	Sb-129	2.78E+07
Cs-136	5.16E+06	Sr-89	9.30E+07
Cs-137	9.47E+06	Sr-90	7.15E+06
I-131	8.82E+07	Sr-91	1.13E+08
I-132	1.27E+08	Sr-92	1.21E+08
I-133	1.82E+08	Tc-99m	1.47E+08
I-134	2.00E+08	Te-127	9.05E+06
I-135	1.70E+08	Te-127m	1.18E+06
Kr-85	9.01E+05	Te-129	2.74E+07
Kr-85m	2.47E+07	Te-129m	4.12E+06
Kr-87	4.78E+07	Te-131m	1.26E+07
Kr-88	6.74E+07	Te-132	1.26E+08
La-140	1.62E+08	Xe-133	1.82E+08
La-141	1.49E+08	Xe-135	5.06E+07
La-142	1.45E+08	Y-90	7.46E+06
Mo-99 <sup>c</sup>	1.68E+08	Y-91	1.19E+08
Nb-95 <sup>c</sup>	1.62E+08	Y-92	1.22E+08
Nd-147	5.95E+07	Y-93	1.39E+08
Np-239	1.68E+09	Zr-95 <sup>c</sup>	1.60E+08
Pr-143	1.39E+08	Zr-97 <sup>c</sup>	1.58E+08

<sup>a</sup>Inventory based on ORIGEN2 calculation of extended-burnup PWR; 51,700 MW-D/MT burnup; 4.7% enriched; 18-month cycle; 81 capacity factor; 88.128 MT core inventory; at end of equilibrium cycle.

<sup>b</sup>Co-58 and Co-60 inventory caused by activation of cobalt impurities in fuel assembly components.

<sup>c</sup>Inventory for these isotopes is the sum of fission product and activation product inventories. Fission product inventory accounts for greater than 95% of the total.

Table B.2.2. Core Inventory Isotopic Ratios: MOX/VO<sub>2</sub> core<sup>a</sup>

Isotope	MOX/VO <sub>2</sub> Ratio	Isotope	MOX/VO <sub>2</sub> Ratio
Am-241	4.85	Pu-238	0.58
Ba-139	0.91	Pu-239	3.03
Ba-140	0.93	Pu-240	3.75
Ce-141	0.94	Pu-241	3.49
Ce-143	0.86	Rb-86	0.50
Ce-144	0.80	Rh-105	1.43
Cm-242	3.54	Ru-103	1.26
Cm-244	1.42	Ru-105	1.44
Co-58 <sup>b</sup>	1.00	Ru-106	1.76
Co-60 <sup>b</sup>	1.00	Sb-127	1.40
Cs-134	0.65	Sb-129	1.26
Cs-136	1.34	Sr-89	0.57
Cs-137	1.04	Sr-90	0.50
I-131	1.10	Sr-91	0.64
I-132	1.04	Sr-92	0.70
I-133	1.00	Tc-99m	0.95
I-134	0.96	Te-127	1.40
I-135	0.92	Te-127m	1.54
Kr-85	0.58	Te-129	1.26
Kr-85m	0.62	Te-129m	1.30
Kr-87	0.57	Te-131m	1.20
Kr-88	0.59	Te-132	1.05
La-140	0.93	Xe-133	0.98
La-141	0.93	Xe-135	1.46
La-142	0.90	Y-90	0.48
Mo-99	0.98	Y-91	0.63
Nb-95	0.85	Y-92	0.71
Nd-147	0.97	Y-93	0.78
Np-239	0.40	Zr-95	0.85
Pr-143	0.86	Zr-97	0.93
Xe-131m	1.20	H-3	1.09
Xe-133m	1.00	Kr-83m	0.58
Xe-135m	1.46	Xe-138	0.91

<sup>a</sup>Ratio based on Table 1.3.4-3 of Ref. 4, unless otherwise noted in text.

<sup>b</sup>Co-58 and Co-60 ratios were set to 1, the PDR-600 reported inventories assume the use of stainless steel cladding which yields MOX/VO<sub>2</sub> ratios of approximately 50; stainless steel cladding is not now anticipated to be used in either existing or advanced reactors.

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Year	Average of Four Large BWRS	Average of Eight Large PWRS
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## References

1. U.S. Nuclear Regulatory Commission, Annual Reports (1988-1992), *Radioactive Materials Released from Nuclear Power Plants*, NUREG/CR-2907, Vols. 9-12
2. Institute of Nuclear Power Operations, 1994 Annual Report.



#### B.4 SPENT FUEL GENERATION CHANGES

*Changes to spent fuel generation rates could result from increased refueling frequency or other factors unique to using mixed oxide fuel. Provide estimates, in number of assemblies per year per reactor, of potential increases in spent fuel generation rates. Also, provide estimates of additional wet and dry storage space, in cubic feet (or number of assemblies), that might be required to accommodate the additional spent fuel.*

Four existing reactor/fuel cycles are identified from Appendix C of the October 25, 1994, draft of Volume I of Phase II of the NE Reactor Pu Disposition Study. In each of these cases, 50 metric tonnes of plutonium is consumed in 30 years or less. The cases identified were:

- CE-ER1 is a 1256 MWe System 80 Combustion Engineering pressurized water reactor with a capacity factor of 0.82,
- CE-ER2 is a 1153 MWe, 3410 Class, Combustion Engineering pressurized water reactor with a capacity factor of 0.75,
- W-ER is a 1150 MWe, 4 loop, Westinghouse pressurized water reactor with a capacity factor of 0.75, and
- GE-ER is a 1155 MWe, BWR-5, General Electric boiling water reactor with a capacity factor of 0.75.

Table B.4.1 provides a comparison of assemblies discharged per year between existing LEU plants and the MOX fueled plants. Also shown in Table B.4.1 is the average number of assemblies discharged per year for the plants shown in Appendix A.

The decay heat rate of discharged fuel assemblies initially charged with weapons-grade plutonium is within a few percent of current LEU discharged fuel. Consequently, there will be no negative impact on the cooling needed for irradiated fuel element storage due to the substitution of MOX fuel for LEU fuel.

Since the number of assemblies discharged per year for most of the MOX cycles is greater than that of the average LEU cycle, the amount of wet or dry storage required for the MOX cycles will be more than for the average of LEU cycles. Some facilities in the United States have applied for and received authorization to hold spent fuel in dry storage at existing reactor sites. This storage of fuel in dry storage casks should not require significant additional buildings nor land space to accommodate storage until a suitable repository is available. None of the plants examined in Table A, due to their young age, have applied for nor received permission to store spent fuel in dry storage casks. Spent fuel is required to be maintained in the spent fuel storage pool for a minimum of 10 years to allow for sufficient cooling. The increased number of MOX spent fuel assemblies shown in Table B.4.1 will therefore need to be held in an existing pool for a 10-year period. This could necessitate an increase in size of the pools to accommodate ten times the number of additional assemblies. However, all of the plants in Appendix A have sufficient fuel pool capacity to accommodate additional assemblies resulting from use of MOX fuel. Additionally, dry storage on-site would alleviate pool crowding until shipment of the spent fuel to a repository.

Table B.4.1. Additional Spent Fuel Generation Rates  
(Values rounded upward—18-month refueling cycle)

Reactor	Assemblies discharged per year	Change from average PWR	Change from average BWR
CE-ER1 (MOX)	81	+33	—
CE-ER2 (MOX)	109	+61	—
W-ER (MOX)	51	+3	—
Average PWR (LEU)	48 <sup>1</sup>	—	—
GE-ER (MOX) (ABWR)	142	—	+15
Average BWR (LEU)	127 <sup>2</sup>	—	—

<sup>1</sup> Table A average for PWR:  $72 \div 1.5 = 48$  assemblies per year.

<sup>2</sup> Table A average for BWR:  $190 \div 1.5 = 126.7$  (127) assemblies per year.

## B. 5 GASEOUS RADIOACTIVE EFFLUENT

*Empirical gaseous effluent data for 12 LWRs with electrical ratings in excess of 1100 MWe for a period of five years is provided in Appendix A and the table below. Provide estimates in the appropriate columns for expected increase of gaseous radioactive effluent, if any, for a reactor using mixed oxide fuel.*

### B.5.1. Background

Reference 1 describes an approach that was used to obtain gaseous and liquid source terms for normal operation of mixed oxide reactors. This study was performed in the 1970s when recycling reactor grade plutonium (with a different isotopic mix than WG-MOX) from LWRs was under consideration. Reference 1 examined the release of fission products from typical large (approximately 3000 MWth) BWRs, PWRs with U-tube steam generators, and PWRs with direct cycle steam generators. This study followed the NRC recommended approach as outlined in Ref. 2.

Reference 2 specifies a calculational methodology for the determination of operational radiological source terms for the purposes of showing compliance to Appendix I of 10CFR Part 50 and to determine the environmental impact of an operating plant. The calculated effluents for a UO<sub>2</sub> plant were determined using the BWR-GALE and the PWR-GALE computer codes as prescribed by Ref. 2. Reactor core nuclide inventory ratios of MOX fuel to UO<sub>2</sub> fuel (in concert with a number of other simplifying assumptions), were applied to the effluents from UO<sub>2</sub> fueled reactors to calculate the MOX effluents. The expected MOX effluents in Ref. 1 were estimated for each reactor type by using MOX/UO<sub>2</sub> ratios and applying them to the quantity of isotopes calculated in the effluent. It is significant to note that the MOX nuclide inventories that were used in this calculation are somewhat different (e.g. different plutonium isotopic feed mixes, different exposure time etc.) from the MOX spent fuel inventory that would be expected using weapons grade MOX. Core inventories for a full MOX core (using weapons grade MOX) are addressed in Section B.2.

### B.5.2. Approach

For this PEIS, it was decided to use a similar approach (as described above) to obtain the gaseous and liquid emissions for a generic light-water reactor burning full MOX fuel. Table A reports the statistics associated with the yearly gaseous and liquid effluent data that are reported by the utilities. This table uses the emissions data from operating plants. It was decided to take advantage of the abundance of measured data and the reported values in this table. Thus, the values in Table A are assumed to constitute an established baseline for the quantification of emissions from large UO<sub>2</sub> fueled light-water reactors.

The baseline emissions for UO<sub>2</sub> plants are adjusted to account for differences (resulting from full MOX use) in two respects: isotopic yields for MOX vs UO<sub>2</sub>, and the possible increase in fission gas in the fuel rod plenum (known as the "gap") that is specific to MOX fuel. The approach employed in this analysis is somewhat similar to the approach in Ref. 1, but makes use of reported annual data from operating reactors and current predicted MOX fuel performance. The approach described generally applies to both the gaseous effluents reported in this section and the liquid effluents that are addressed in Section B.6. After adjusting UO<sub>2</sub> emissions for a MOX plant, the expected emissions from a MOX plant can be compared to the statistical variations that are reported in Table A. This comparison will enable one to see how predicted MOX plant emissions compare to the span in emissions for UO<sub>2</sub> plants.

### B.5.3 Major Assumptions

Simply stated, radioactive effluents find their origins in nuclear fuel and its related performance and interact with the medium (primary coolant) that surrounds it. The first step of the approach is to infer the concentration of radionuclides present in the primary coolant. This approach necessitates that an important assumption be made as to the performance of MOX fuel rods. *It is assumed that the number of MOX fuel pins that leak is equivalent to the number of UO<sub>2</sub> fuel pins that leak.*



MOX/ UO <sub>2</sub> Ratio <sup>b</sup>	MOX Average <sup>c</sup>	Δg (MOX-UO <sub>2</sub> )
1.0	3.59	0.0
0.58	20.48	-0.66
0.62	24.32	0.83
0.59	26.43	-0.39
1.20	10.72	5.37
0.98	1,149.04	416.95
1.00	10.79	5.33
1.46	162.31	98.71
1.02	118.76	53.52

## B.6 LIQUID RADIOACTIVE EFFLUENT

*Empirical liquid effluent data for 12 PWRs with electrical ratings in excess of 1100 MWe for a typical year is provided in the table below. (Isotopes for all effluents are listed in Appendix A, Commercial Reactor Operational Characteristics.) Provide estimates in the appropriate columns for expected increases of liquid radioactive effluent, if any, for a reactor using mixed oxide fuel.*

The approach for obtaining liquid effluents for Xe-133 and tritium for the MOX plant is described in Section B.5. However, Fe-55 and Co-58 are activation products and are treated appropriately. The resulting predicted effluents are shown in Table B.6 below. The table shows that the liquid effluents expected from a MOX plant are quite similar to a UO<sub>2</sub> reactor. This conclusion is sensible given the assumption that MOX fuel performs similarly to UO<sub>2</sub> fuel (see discussion in Section B.5).

The vendor reports discuss the possibility of plutonium carryover in waste streams from an existing plant. Based on the information in Refs. 4 and 5, it is expected that the use of MOX fuel will not result in the presence of plutonium in a liquid or gaseous waste stream during normal operations.

As was the case with gaseous emissions, the calculated core inventory differences (due to the use of MOX fuel) and the factor of 1.67 have a very small effect on liquid emissions. It is concluded that the liquid emissions of a plant using MOX fuel are very similar to a plant using UO<sub>2</sub> fuel.

### References

1. U.S. Nuclear Regulatory Commission, *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors*, NUREG-002, Vol. 3, August 1976.
2. U.S. Nuclear Regulatory Commission, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water-Cooled Power Reactors*, Regulatory Guide 1.112, May 1977.
3. U.S. Nuclear Regulatory Commission, *Radioactive Materials Released from Nuclear Power Plants*, NUREG/CR-2907, Vol. 9, Annual Report 1988.
4. Westinghouse Electric Corporation, *Plutonium Disposition in Existing Pressurized Water Reactors*, DOE/SF/19683-6, June 1, 1994.
5. GE Nuclear Energy, *Study of Plutonium Disposition Using Existing GE Boiling Water Reactors*, NEDO-32361, June 1, 1994.
6. Westinghouse Electric Corporation, *Performance of MOX Fuel in Light-Water Reactors*, Letter from M. L. Travis to S. E. Fisher, NDP-95-004, February 23, 1995.



Table B.6. Additional Liquid Radioactive Effluent (Ci/year)

Isotope	UO <sub>2</sub> Avg <sup>a</sup>	UO <sub>2</sub> Low <sup>a</sup>	UO <sub>2</sub> High <sup>a</sup>	MOX/UO <sub>2</sub> Ratio <sup>b</sup>	MOX Average <sup>c</sup>	Delta Avg (MOX-UO <sub>2</sub> )
Fe-55	0.15	0.0	0.73	1.00	0.15	0.0
Co-58	0.17	0.0	0.65	1.00	0.17	0.0
Xe-133	2.37	0.0	23.49	0.98	3.88	1.51
Tritium	385.85	0.53	938.25	1.09	702.36	316.51

<sup>a</sup>Yearly emissions extracted from Table A.

<sup>b</sup>Ratio = MOX system isotopic inventory/UO<sub>2</sub> system inventory, from Table B.2.2.

<sup>c</sup>MOX value = (UO<sub>2</sub> Avg) (MOX/UO<sub>2</sub> ratio) (1.67).

## B.7 ADDITIONAL WORKER EXPOSURE

*Empirical exposure data for typical job categories for 12 LWRs with electrical ratings in excess of 1100 MWe for a period of five years is provided in the table below. The data in the first table is representative of refueling operations only. The data in the second table is representative of all plant operations. Provide estimates in the appropriate columns for expected increases of exposure in person-rem, if any, for a reactor using mixed oxide fuel. Any additional integrated exposure estimates should be attributable to (1) additional employees required for mixed oxide fuel operations, and (2) the additional or different activities required to be executed.*

One design criterion of current LWRs has been the control of radiological impacts during all phases of operation, including fresh fuel handling, refueling and storage activities. Current LWRs are licensed and operated in accordance with the requirements of 10CFR20, Standards for Protection Against Radiation. These requirements include (1) establishment of an operational radiation protection program which achieves occupational radiation doses as low as reasonably achievable (ALARA), and (2) administrative controls to limit individual doses to less than:

- 5 rem/year total effective dose equivalent;
- 50 rem/year to individual organs;
- 15 rem/year dose equivalent to the eye; and
- 50 rem/year dose equivalent to the skin and extremities.

Typical occupational doses to LWR workers have been compiled from exposure rates reported in the references in Appendix A for operating LWRs. The occupational exposures for LWRs are presented in Tables B.7.1 and B.7.2. These doses are the typical annual averages based on normal expected refueling cycles in an LWR. Operation of an LWR with MOX fuel may slightly increase the occupational exposures because of exposures encountered during new fuel storage, inspection, and movement. The additional exposures projected from the use of MOX fuel (delta) are provided in the tables. Reference 1 reported slight additional worker exposure—0.6 person-rem/year for refueling and 1.0 person-rem/year for normal operation. No other breakdown by job category was given.

### References

1. "Analysis of Existing ABB-CE LWRs for the Disposition of Weapons-Grade Plutonium," DOE Plutonium Disposition Study, Combustion Engineering, Inc., Windsor, Connecticut, June 1, 1994.
2. "Study of Plutonium Disposition Using Existing GE BWRs," NEDO-3232361, GE Nuclear Energy, June 1, 1994.
3. "Plutonium Disposition in Existing PWRs," DOE/SF/19683-6, Westinghouse Electric Corporation, June 1, 1994.



Table B.7.1. Additional Worker Exposure (Person-rem) Refueling Operations

Job Category	Avg.	Low	High	Plant Using Mixed Oxide Fuel
Maintenance	21.44	4.32	51.70	
Operations	2.59	0.01	8.52	
Health Physics	4.14	0.77	15.13	
Supervisory	0.52	0.0	1.68	
Engineering	3.01	0.11	12.45	
Total	31.70	5.21	89.48	0.6*

\*MOX fuel movement

Table B.7.2. Additional Worker Exposure (Person-rem) All Operations

Job Category	Avg.	Low	High	Plant Using Mixed Oxide Fuel
Maintenance	201.15	74.17	511.40	
Operations	28.00	4.84	62.96	
Health Physics	44.83	15.10	89.62	
Supervisory	5.77	0.05	29.80	
Engineering	23.91	5.86	51.14	
Total	303.66	100.02	744.92	1.0*

\*Total for all fresh fuel storage activities

## B.8 CHANGES TO ACCIDENT BASIS

*Identify any additional accident scenarios due to mixed oxide fuel use or activities that are not within the envelope of the accepted accident basis for currently operating commercial PWRs, the probability of occurrence, the quantity of radiological/hazardous materials at risk, the quantities of material dispersed in each scenario, and accident consequences. Discuss qualitative mitigation factors.*

### B.8.1. Background

For existing reactors, accidents are classified in environmental reports as Class 1 through Class 8 accidents as prescribed in NRC Regulatory Guide 4.2, Appendix I. These classes typically encompass events such as a rupture of the waste gas decay tank, spent fuel handling accidents, steam generator tube ruptures, and both small and large break loss of coolant accidents (LOCAs). The accident spectrums discussed in this Section are broken into two major categories: nonsevere accidents and severe accidents. Before beginning the detailed discussion of accident categories, it should be noted that no independent analyses have been performed at ORNL to investigate the fission product releases associated with the use of MOX fuel. The following discussion is primarily based upon the information supplied by the vendors for the Plutonium Disposition Studies.

With respect to the consideration of non-severe accidents for the WNP-1 plant [utilized as a New Production Reactor (NPR)], Ref. 2 states "*The non-severe accidents encompass an accident spectrum that includes events that are not expected in the life of the reactor in addition to events that might be expected during the life of the reactor.*" It is noteworthy that there is overlap between accidents evaluated in environmental reports and accidents identified as "design-basis." As an example, a large break LOCA is an accident that is discussed in environmental reports and also addressed in FSARs as a design basis accident. In this Section, the emphasis is on accident categories found in environmental reports. The term non-severe accidents is used to describe these events.

Severe accidents are distinguishable from non-severe accidents in that they involve substantial physical deterioration (including melting) of the fuel. In addition, severe accidents typically involve a compromise of the containment structure or a significant bypass of the containment. Thus, the source terms from these accidents are generally large as compared to non-severe accidents, but the probabilities associated with severe accidents are relatively small.

In the NPR program, the WNP-1 plant (a 3800 MWth PWR) was analyzed as a potential candidate for the production of tritium [1]. A significant effort was made to identify both the non-severe accidents and the severe accident spectrum for this reactor. In addition, consideration was given to the accident issues specifically associated with tritium production in the WNP-1. References 1 and 2 provide a description of these analyses. The discussion of the accident basis provided here makes considerable use of the results contained in Refs. 1 and 2 for both non-severe (termed design basis in the Reference) and severe accidents. These references are also compared against the RPT design basis for the WNP-1 and the FSAR design basis for the WNP-1. The purpose of this comparison is to provide a clear understanding of how the accident basis for the WNP-1 compares to the design basis for the WNP-1 and the FSAR design basis for the WNP-1.

Reference 2 also provides the isotopic source terms for each of the accidents listed in Table B.8.2. In Appendix 1 of Ref. 1, three accidents were selected from the thirteen (listed in Table B.8.2) based on the severity of their consequences (exposures to workers and offsite personnel). The accidents selected were a waste gas decay tank rupture, a steam generator tube rupture, and a large break LOCA. The source terms associated with each of these three accidents are shown in Table B.8.3. The source terms listed in this table were extracted from Appendix 1 of Ref. 1 and scaled from a power level of 3800 MW to a power level of 3404 MW. The scaling factor was taken to be a ratio (3404/3800) of the power level.

For this study, the source terms in Table B.8.3 for the WNP-1 operating as an NPR have been compared to the source terms identified in environmental reports at two commercial plants. The Vogtle and Waterford reactor environmental reports (Refs. 3 and 4) were utilized for this comparison. Both of these reactors are in the 3400 MW range and are representative of large PWRs (WNP-1 has a slightly larger rating, 3800 MWth).

For the gas waste tank rupture accident, there is general agreement in source terms between Waterford and WNP-1 (operating as an NPR). However, the Vogtle plant shows a significantly smaller source term than the other two reactors. For the steam generator tube rupture accident, the comparison is mottled between the three plants. Agreement between some of the isotopes in the source terms is found, whereas others exhibit significant differences. Differences also exist for the large break LOCA. There is reasonable agreement for the noble gas release between the Vogtle and Waterford plants, but the iodine release for the Waterford plant is quoted to be substantially less than the Vogtle plant. The WNP-1 plant indicates a release that is less than the other two plants for a large break LOCA.

Examination of environmental reports for UO<sub>2</sub> fueled commercial reactors shows that the isotopic source terms vary for the same accident at different plants. Differences in plant equipment and design are the suspected cause of these differences. Assumptions in the plant specific analyses also contribute to differences, even though Regulatory Guide 4.2 is quite prescriptive.

The major objective of the information in this submittal is to identify source term differences between a MOX and UO<sub>2</sub> fueled reactor. Thus, it is thought that the absolute differences in plant to plant source terms are *not* an overriding consideration at this stage of the analysis. It is concluded that Tables B.8.1 through B.8.3 provide an adequate baseline that characterizes the non-severe accidents for a core fueled by low enriched uranium.

### B.8.3. Non-Severe Accident Considerations Resulting from MOX Fuel

A review of the vendor reports for the use of MOX in existing reactors suggests that there are several considerations in regards to source terms. In Ref. 5, there is a Section on fuel performance that discusses the expected effects MOX would have on fuel rod performance. The information shows that at the end of the equilibrium cycle (at high burnups; 30,000 to 40,000 MW-D/MT), fission gas release to the plenum of the fuel rod is increased for MOX as compared to UO<sub>2</sub> fuel. This causes a corresponding increase in the internal gas pressure of the fuel rod. The term "gap release" is normally used in the discussion of accidental releases. This discussion assumes that "plenum release" and "gap release" are equivalent terms.

Reference 5 suggests that a number of strategies can be used to minimize the increase of fission gases in the fuel. Appropriate fuel manufacturing techniques in tandem with core management strategies could help alleviate the undesirable effect. However, given the state of knowledge at hand, it is conservative and appropriate to consider the effects of increased gap activity and its impact on non-severe accident source terms. This issue is not of significant concern for severe accidents because these accidents already postulate significant fuel damage, breach of cladding, and release of the bulk of the fission gasses.

Figure 2.1.1.2-6 of Ref. 5 is a plot of fission gas release as a function of rod burnup. This information is given for UO<sub>2</sub> fuel, homogeneous MOX fuel, and inhomogeneous MOX fuel. For most burnup levels, inhomogeneous MOX shows the highest fission gas release, followed by homogeneous MOX, and then by 4.6% enriched UO<sub>2</sub> fuel.





Containment pressurization is also driven by the energy released by decay heating and by the gases evolved from core-concrete interaction. Again, because of the similarity of the decay heating process, the timing of containment failure by overpressure should not vary significantly from that calculated for the UO<sub>2</sub>-fueled core [7].

With respect to equipment survivability, most equipment will be subjected to a lower integrated dose because of the existence of more long-lived isotopes in the MOX core fission product spectrum. Nevertheless, the MOX core produces approximately 30% more of the Cs<sup>136</sup> isotope [See Table B.2.2]. Cesium is soluble in water and would collect in containment sumps and be transported via containment sprays. The Cs<sup>136</sup> isotope emits a high-energy gamma, which may require locally increased shielding or extended equipment qualification for some safety systems [7]. This, of course, would be determined and appropriate action would be taken prior to plant operation with a MOX core.

In summary, the integrated decay heat during the period prior to the calculated time of containment failure is approximately equal for the two fuel designs. Accordingly, the overall plant thermodynamic response will be about the same under severe accident conditions, and the containment failure probabilities for the MOX and UO<sub>2</sub>-fueled cores will be similar.

NUREG-1465, issued by the U.S. Nuclear Regulatory Commission Staff during February 1995, provides revised accident source terms incorporating severe accident insights developed over the last 15 years with respect to light-water reactor (LWR) severe accidents and the behavior of the fission products that would be released into containment. The revised source terms are presented as separate tabulations for BWRs and PWRs. They provide the magnitude and composition of fission product releases into containment for each of the various phases (gap, early in-vessel, ex-vessel, and late in-vessel) of a severe accident. The release fractions associated with the NUREG-1465 source terms are intended to be representative or typical, rather than conservative or bounding values [11].

The NRC staff plans to use the NUREG-1465 source terms for the Westinghouse AP600 and General Electric Simplified Boiling Water Reactor (SBWR) designs and has an effort underway to allow existing light-water reactor licensees to voluntarily use the revised source terms as well [11]. Thus, it seems prudent to determine if the release fractions from the core to the environment listed in Table B.8.5 are compatible with the release fractions from the core to the containment specified by NUREG-1465.

The total releases (all severe accident phases) into containment for the PWR expressed as fractions of core inventory are:

Noble gasses	Xe-Kr	1.0000
Halogens	I-Br	0.7500
Alkali metals	Cs-Rb	0.7500
Tellurium group	Te-Sb-Se	0.3050
Strontium	Sr	0.1200
Noble metals	Ru	0.0050
Lanthanides	La	0.0052
Cerium group	Ce-Pu-Np	0.0055
Barium	Ba	0.1200

from Table 3.13 of NUREG-1465.\* Based upon these release fractions from core to containment, the release fractions from containment to environment are derived from the Table B.8.5 entries and are listed in Table B.8.6.

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\* Total releases for the BWR differ only for the halogens and alkali metals, with values of 0.70 instead of 0.75.

As an example of how the entries in Table B.8.6 are derived, Table B.8.5 indicates that release category RC1 involves a transfer of 40% of the  $\text{I}^{\text{m}}\text{Br}$  in the core to environment while NUREG-1465 states that 75% of the halogens should be considered to escape from core to containment. The corresponding release from containment to environment is then  $0.40/0.75$  or  $0.533$ , as indicated in Table B.8.6. Since release category RC1 involves containment bypass (See Table B.8.4), a release of slightly more than half of the containment inventory to environment seems reasonable.

Containment-to-environment release fractions of  $5 \times 10^{-4}$  or less have been omitted from Table B.8.6 for clarity. For release category RC1, the containment release fraction derived for the noble metals (Ru) is greater than one, which is impossible. This means that the value of  $1.4 \times 10^{-2}$  listed in Table B.8.5 for Ru release category RC1 is conservatively high when compared to the NUREG-1465 source term. Specifically, NUREG-1465 provides that only  $5 \times 10^{-3}$  of the noble metals should be considered to be released from the core to the containment.

All of the other entries in Table B.8.6 seem reasonable. The largest containment release fractions are associated with release categories RC1, RC1A, RC3, and RC4, which involve either containment bypass or containment isolation failure with the containment sprays failed. The smallest release fractions occur for RC8, which involves an intact containment. The release fractions for category RC7 are also less than 0.001 for all fission product groups except noble gases. Release category RC7 represents late containment failure with sprays operating so that the containment atmosphere is well-scrubbed when failure occurs.

#### **B.8.7. Severe Accident Source Terms for a MOX Fueled Plant**

As explained in Section B.8.6, the definitions of the release categories, the associated frequencies per reactor year, and the fission product group release fractions derived for the existing reactors using  $\text{UO}_2$  fuel are expected to be unchanged if operation is shifted to the use of MOX fuel. This conclusion is based upon the analyses that have been performed by the reactor vendors for the limiting transients that have been identified for the plutonium disposition core designs [7, 8, 9].

The fission product inventories in the core at the time of shutdown at the inception of an accident will differ if MOX fuel is substituted for the normally used  $\text{UO}_2$  fuel. While some fission products would exist in greater quantities, some would have smaller inventories. Table B.2.2 provides the ratios of the MOX inventory to the  $\text{UO}_2$  inventory for the fission products that are of interest when performing MACCS code consequence calculations.

For the purpose of identifying the changes in accident consequences associated with the use of MOX fuel, the MOX core fission product inventories can be obtained by multiplying the typical  $\text{UO}_2$  fueled light-water reactor inventories listed in Table B.2.1 by the individual isotope ratios given in Table B.2.2. Then the MACCS calculations can be run employing the release frequencies and release fractions listed in Table B.8.5. This will produce the accident consequences for the use of MOX, which can then be compared with the consequences predicted by the MACCS code for the  $\text{UO}_2$  initial inventories (Table B.2.1).

#### **References**

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3. Georgia Power, *Vogtle Electric Generating Plant Unit 1 and Unit 2: Applicants Environmental Report, Operating License Stage*, Vol. 2, April 1984.

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Table B.8.1. Non-Severe Event Frequency Categorization

Event	Frequency Categorization	Range of Probability
Radwaste system failure	Moderate frequency	$10^{-1}$ to $2 \times 10^{-2}$
Steam generator tube rupture	Infrequent accident	$2 \times 10^{-2}$ to $2 \times 10^{-3}$
Refueling accident	Infrequent accident	$2 \times 10^{-2}$ to $2 \times 10^{-3}$
Loss-of-coolant accident	Limiting fault	$< 2 \times 10^{-3}$
Control rod ejection accident	Limiting fault	$< 2 \times 10^{-3}$
Main steam line break	Limiting fault	$< 2 \times 10^{-3}$

Source: Reference 2

Table B.8.2. Non-Severe Events Under Consideration

Accident	Frequency Category	Description
LWR1	Infrequent	Equipment leakage or malfunction resulting in release of 25% of the normal inventory of a waste-gas decay tank
LWR2	Infrequent	Equipment leakage or malfunction resulting in release of 100% of the normal inventory of a waste-gas decay tank
LWR3	Infrequent	Off-design transients that induce fuel failures above those expected and steam generator leak combined with station blackout
LWR4	Infrequent	Steam generator tube rupture
LWR5	Infrequent	Refueling accident – fuel assembly drop
LWR6	Infrequent	Refueling accident – heavy object drop onto fuel in core
LWR7	Infrequent	Spent fuel handling accident – fuel assembly drop into fuel storage pool
LWR8	Infrequent	Spent fuel handling accident – heavy object drop onto fuel rack
LWR9	Limiting	Loss-of-coolant – small pipe break
LWR10	Limiting	Loss-of-coolant – large pipe break
LWR11	Limiting	Loss-of-coolant – break in letdown line from primary system that penetrates the containment
LWR12	Limiting	Rod ejection accident
LWR13	Limiting	Steam-line break outside containment

Source: Reference 2

Table B.8.3. Selected Non-Severe Events and Respective Source Terms UO<sub>2</sub> Fueled Reactor<sup>a</sup>

Radio-nuclide	LWR2 (Curies) 100% Waste Gas Decay Tank Rupture		LWR4 (Curies) Steam Generator Tube Rupture		LWR10 (Curies) Large Break LOCA	
	0-2 hours <sup>b</sup>	Entire <sup>b</sup>	0-2 hours <sup>b</sup>	Entire <sup>b</sup>	0-2 hours <sup>b</sup>	Entire <sup>b</sup>
Kr-83m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.85E-02	8.06E-01
Kr-85m	2.60E+02	2.60E+02	2.78E+01	2.78E+01	4.39E-01	6.36E+00
Kr-85	1.61E+04	1.61E+04	4.03E+01	4.03E+01	2.06E-01	4.66E+01
Kr-87	2.15E+02	2.15E+02	2.78E+01	2.78E+01	3.31E-01	2.33E+00
Kr-88	4.39E+02	4.39E+02	5.20E+01	5.20E+01	8.96E-01	9.85E+00
I-131	3.58E+00	3.58E+00	4.93E+00	4.93E+00	1.70E+00	6.81E+01
I-132	1.43E+00	1.43E+00	2.33E+01	2.33E+01	2.42E-01	1.43E+00
I-133	2.42E+00	2.42E+00	1.88E+01	1.88E+01	1.25E+00	1.97E+01
I-134	1.79E+00	1.79E+00	3.31E+01	3.31E+01	2.06E-01	1.16E+00
I-135	2.15E+00	2.15E+00	2.87E+01	2.87E+01	6.45E-01	6.36E+00
Xe-131m	2.51E+04	2.51E+04	1.79E+02	1.79E+02	1.16E-01	8.96E+00
Xe-133m	4.48E+02	4.48E+02	1.34E+01	1.34E+01	2.78E-01	1.52E+01
Xe-133	2.96E+04	2.96E+04	4.39E+02	4.39E+02	1.43E+01	1.34E+03
Xe-135m	3.14E+02	3.14E+02	1.34E+01	1.34E+01	2.15E-01	4.66E+00
Xe-135	3.40E+03	3.40E+03	2.51E+02	2.51E+02	9.85E-01	2.78E+01
Xe-138	1.34E+02	1.34E+02	1.79E+01	1.79E+01	1.07E-01	5.20E-01

<sup>a</sup>Tritium, which was significant due to NPR targets is not included in this table; Refs. 3 and 4 do not indicate tritium to be of concern in existing LWRs for non-severe accidents

<sup>b</sup>All source terms from Ref. 1 were scaled by applying a ratio to the source terms to 3404 MW from the original power level of 3800 MW.

Source: Reference 1. The original nomenclature included an A at the end of each sequence title; this was a reference to the tritium barrier concept for NPR.

Table B.8.4. Release Categories for the LWR Severe Accident Analysis

Release Category	Description for LWR
RC1	Reactor containment bypass via an interfacing system's loss-of-coolant accident (LOCA) resulting in a low-pressure decay heat removal system pipe rupture in the General Services Building (GSB) with the ventilation system fire dampers failed open.
RC1A	Reactor containment bypass via an interfacing system's LOCA resulting in rupture of a low-pressure decay heat removal system pipe in the GSB with the ventilation system fire dampers successfully closed.
RC2	Reactor containment bypass via an interfacing system's LOCA resulting in failure of a low-pressure decay heat removal system seal, thus flooding the decay heat removal system sump in the GSB with the ventilation system fire dampers failed open.
RC2A	Reactor containment bypass via an interfacing system's LOCA resulting in failure of a low-pressure decay heat removal system seal, thus flooding the decay heat removal system sump in the GSB with the ventilation system fire dampers successfully closed.
RC3	Reactor containment isolation failure at high energy with containment sprays failed.
RC3A	Reactor containment isolation failure at high energy with containment sprays functioning.
RC4	Reactor containment isolation failure at low energy with containment sprays failed.
RC4A	Reactor containment isolation failure at low energy with containment sprays functioning.
RC5	Early reactor containment failure with containment sprays failed.
RC5A	Early reactor containment failure with containment sprays functioning.
RC6	Late reactor containment failure with containment sprays failed.
RC7	Late reactor containment failure with containment sprays functioning.
RC8	Reactor containment intact.



MOX fuel would have less depleted uranium, but more neutron shielding. Thus, the cask for transporting MPCs containing spent MOX fuel would be conceptually the same as that for conventional LEU fuel, although the dimensions (thicknesses) of the gamma and neutron shielding components would be different. The other potential difference will be in the MPCs basket design, because of differences between conventional and MOX spent fuel assembly reactivities and their impact on the neutron absorber requirements for criticality safety.”

Judging from the Westinghouse commentary, it appears that either an existing shipping cask design could be modified or a new design could be created to achieve the level of safety traditionally provided for the shipment of spent LEU fuel.

Reactor (Number of reactor units required)	Cycle length (mo)	Assemblies charged per cycle (total all units)	Curies per assembly <sup>a</sup>	Shipments per year <sup>c</sup>	Curies per shipment	Assemblies needed to dispose 50 MFC Pu <sup>d</sup>
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**Table B.9.2. Spent MOX Fuel Shipments**  
(All values rounded upward)

Reactor	Assemblies discharged per year <sup>a</sup>	Annual spent fuel shipments needed (rail) <sup>b</sup>	Annual spent fuel shipments needed (truck) <sup>c</sup>
CE-ER1 (2 units)	162	8	41
CE-ER2 (2 units)	218	11	55
GE-ER (3 units)	426	11	48
W-ER (3 units)	153	8	39
Avg. BWR (LEU) (per unit)	127	4	15
Avg. PWR (LEU) (per unit)	48	3	12

<sup>a</sup>Assemblies discharged per year is the total assemblies discharged per year from all units.

<sup>b</sup>Rail shipments assume utilization of the Large Multi-Purpose Canister (MPC) System being designed by the Civilian Radioactive Waste Management System (CRWMS) M&O Contractor, that accommodates 21 PWR or 40 BWR in a 125 ton rail cask. Also under development is a Medium MPC, weighing 75 tons, that holds 12 PWR or 24 BWR assemblies for utilities that cannot handle the 125-ton cask system. The MPC design will have to be re-evaluated using the MOX spent fuel characteristics.

<sup>c</sup>A number of legal-weight truck casks are under consideration by the CRWMS M&O. These casks include the NAC-LWT (1 PWR or 2 BWR spent fuel assemblies) and the GA-4/9 (4 PWR or 9 BWR assemblies). In this study, the larger capacity GA-4/9 is assumed.

